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REVIEW ARTICLE

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Liquid metal thermal hydraulics R&D at European scale: achievements and prospects

Abstract A significant role for a future nuclear carbon-free energy production is attributed to fast reactors, mostly employing a liquid metal as a coolant. This paper summarizes the efforts that have been undertaken in collaborative projects sponsored by the European Commission in the past 20 years in the fields of liquid-metal heat transfer modeling, fuel assembly and core thermal hydraulics, pool and system thermal hydraulics, and establishment of best practice guidelines and verification, validation, and uncertainty quantification (UQ). The achievements in these fields will be presented along with the prospects on topics which will be studied collaboratively in Europe in the years to come. These prospects include further development of heat transfer models for applied computational fluid dynamics (CFD), further analysis of the consequences of fuel assembly blockages on coolant flow and temperature, analysis of the thermal hydraulic effects in deformed fuel assemblies, extended validation of three-dimensional pool thermal hydraulic CFD models, and further development and validation of multi-scale system thermal hydraulic methods.

Keywords liquid metal, thermal hydraulics, Europe

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1 Introduction

Exact predictions on the future energy landscape are hard to provide. However, all recent major studies [1–7] indicate that there is no simple solution for the climate change issues we are facing and that nuclear energy has to continue to play a significant role in the future carbon-free energy production since the carbon-dioxide lifecycle emissions of nuclear are among the lowest of the electricity supply technologies [8]. Even though the relative contribution of nuclear energy in the total energy mix might decrease, the rising global energy demand requests an increase of nuclear energy production in absolute terms. With the current uranium reserves [9], the global fleet of nuclear reactors can be fed for one or maybe two centuries. This indicates a clear need to increase the sustainability of the nuclear fuel cycle which can be achieved by switching from the once-through fuel cycle currently used in water cooled reactors with a thermal neutron spectrum to a closed fuel cycle using fast spectrum reactors. Of the fast reactor designs around the world, the liquid metal cooled designs are the most mature. In fact, many such reactors have already been operated since the dawn of nuclear.

Worldwide, various developments [10] are worthwhile mentioning. First of all, the BN800 sodium-cooled reactor in Russia provided first electricity to the grid in 2015. Following this deployment, Russia is now designing the BN1200 sodium-cooled and the BREST-300 (Russian: Fast Reactor with Lead Cooling) lead-cooled reactors. In India, the construction and commissioning of the PFBR (Prototype Fast Breeder Reactor) sodium-cooled reactor is in an advanced stage. In the US, the new advanced VTR (Versatile Test Reactor) sodium-cooled test reactor is being developed, and recently the development of the Natrium™ reactor has been announced. China operates the small CEFR (China Experimental Fast Reactor) sodium-cooled test reactor and is now constructing the CFR600 (China Fast Reactor 600 MW_e) sodium-cooled demonstration reactor, while the design work on the lead-cooled CLEAR

(China Lead-based Reactor) reactor series is ongoing. In Europe, the UK has serious plans to deploy an advanced fast reactor. Mid 2020, the Westinghouse lead-cooled fast reactor design was selected for further development in the UK context. Other European examples in the design stage are the lead-cooled ALFRED (Advanced Lead Fast Reactor European Demonstrator) reactor envisaged to be constructed in Romania and the small lead-cooled SEALER (Swedish Advanced Lead Reactor) reactor concepts to serve small remote communities or to provide district and/or process heat and electricity to the grid. Apart from that, in Belgium pre-licensing work is ongoing for the lead-bismuth cooled MYRRHA (Multi-purpose Hybrid Research Reactor for High-tech Applications) test reactor.

In every nuclear reactor, core cooling is of utmost importance, both in operational conditions, as well as in accident conditions. Therefore, thermal hydraulic assessment of such reactors is very important. To this aspect, experiments and numerical simulation have to go hand in hand [11]. This paper will summarize the achievements and prospects with respect to liquid metal nuclear thermal hydraulics in the context of the developments at a European collaborative level. The work reported has been performed in various collaborative projects sponsored by the European Commission. First, achievements and prospects related to liquid-metal heat transfer modeling, core, pool, and system thermal hydraulics will be described. After that, verification and validation (V&V) and uncertainty quantification (UQ) will be addressed. Finally, a short summary and an outlook will be provided. Since this paper mentions many experimental facilities in use all over Europe supporting design and safety analyses of liquid metal fast reactors, a non-exhaustive list of facilities is provided in Appendix A for the reader to consult. For further information on these test facilities, the reader is recommended to consult the database from the IAEA [12].

2 European collaborative projects

In Europe, a lot of development takes place in collaborative projects within various framework programs sponsored by the European Commission. The European Commission stimulates coordinated pan-European research through multi-year framework programs in which funding from the European Commission is supplemented by national or institutional/company funding. Figure 1 shows the past and present European collaborative projects related to liquid metal thermal hydraulics research. The columns indicate the framework program to which a collaborative project belongs. The rows indicate the research and development (R&D) field. The specific sodium fast reactor (SFR) related research, including thermal hydraulics, is indicated in the SFR row. Likewise, the research for lead fast reactors (LFR) and lead bismuth-cooled accelerator driven systems (ADS) are indicated in the second and third row. The 4th row is dedicated to projects focusing on (liquid metal) thermal hydraulics. And finally, the last row shows two projects related dedicated to the establishment of liquid metal facility networks. Figure 1 clearly indicates the consistent and significant support of the European Commission to the field of liquid metal fast reactors and specifically liquid metal thermal hydraulics. At the moment of writing, the first projects of Horizon 2020 are finished, while the last projects of this framework program are starting. Specifically, in the field of liquid metal thermal hydraulics, the projects PATRICIA (Partitioning and Transmuter Research Initiative in a Collaborative Innovation Action) and PASCAL (Proof of Augmented Safety Conditions in Advanced Liquid-metal-cooled systems) are commenced.

For the interested reader, Table 1 provides more information on the collaborative European projects mentioned in Fig. 1, including information on the duration of

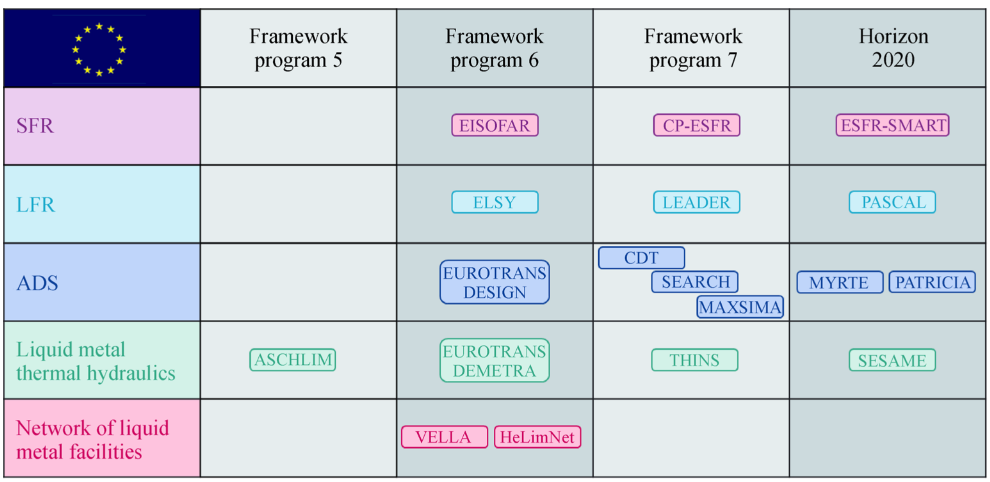

Fig. 1 Liquid metal (thermal hydraulics) related projects sponsored by the European Commission.

Table 1 Collaborative projects in the field of liquid metal thermal hydraulics sponsored by the European Commission

Project	Duartion	EC Budget (M€)	Coordinator (Country)	Topic
ASCHLIM	2002	0.1	SCK (Belgium)	Assessment of applicability of computational fluid dynamics (CFD) for heavy liquid metals
EUROTRANS DESIGN	2005–2010	6.1	SCK (Belgium)	Design of a proton accelerator driven system
EUROTRANS DEMETRA	2005–2010	5.3	KIT (Germany)	Development of materials and liquid metal technologies
VELLA	2006–2009	0	ENEA (Italy)	Virtual European laboratory for heavy liquid metal technology
ELSY	2006–2010	2.9	ANN (Italy)	Demonstration of feasibility of pure lead cooled nuclear reactor
EISOFAR	2007–2008	0.3	CEA (France)	European roadmap for SFR technology
CDT	2009–2012	2.0	SCK (Belgium)	Centralization of research for transmutation
CP-ESFR	2009–2013	5.8	CEA (France)	Development of a large Generation IV sodium fast reactor
HeLimNet	2010–2012	0.5	CNR (Italy)	Harmonizing research in the field of heavy liquid metal technologies
LEADER	2010–2013	3.0	ANN (Italy)	Conceptual design of a European lead fast reactor demonstrator
THINS	2010–2015	5.9	KIT (Germany)	Thermal hydraulics research for all Generation IV reactors
SEARCH	2011–2015	5.7	SCK (Belgium)	Research supporting licensing process of MYRRHA
MAXSIMA	2012–2018	5.5	SCK (Belgium)	Supporting safety assessment of MYRRHA
MYRTE	2015–2019	12.0	SCK (Belgium)	Research to support the demonstration of transmutation
SESAME	2015–2019	5.2	ENEA (Italy) NRG (Netherlands)	Thermal hydraulics safety methods for liquid metal fast reactors
ESFR-SMART	2017–2021	5.0	PSI (Switzerland)	Enhancing safety of Generation IV sodium fast reactors
PASCAL	2020–2024	3.8	ENEA (Italy)	Safety research for innovative heavy liquid metal cooled reactors
PATRICIA	2020–2024	6.5	SCK (Belgium)	Safety related research supporting licensing of MYRRHA

the projects, the budget sponsored by the European Commission (in many cases supplemented with about the same budget from company or national research programs), the coordinating institute and its country of origin, and a short description of the topic. Most projects involve about 10–30 partner institutes from all over Europe bringing together a pan-European pool of experts in a specific field of expertise.

2.1 Heat transfer modeling

The challenge with respect to modeling of heat transfer in liquid metals is that in these fluids typically there is a much larger conductive contribution to heat transfer than in conventional fluids like water and air for which heat transfer models have been developed originally. This leads to the need for developing new heat transfer correlations to be used in back-of-the-envelope calculations and in system thermal hydraulic codes, but also to the need for developing new turbulent heat transfer models for CFD applications.

With respect to correlations, excellent work has been done in Europe in the past 15 years in collecting, assessing, and evaluating available data in, e.g. [13–15]. Their efforts are serving as input for comprehensive data collections from the OECD [16] and the IAEA [17] that are recommended sources of information to be used for property data and correlations. In 2021, these reports are expected to be complemented by a comprehensive IAEA technical document on sodium physical properties and correlations. Interested readers looking for heat transfer or other correlations with various scopes of applications are strongly recommended to download these extensive documents (Refs.[16,17]) to select the appropriate correlations from these well established and recognized databases.

With respect to turbulent heat transfer model development for CFD applications, the current state-of-the-art models are summarized in Refs. [18,19]. From these publications which describe and assess the various options in detail including validation and applicability, it is clear that Algebraic Heat Flux Models (AHFMs) provide a promising route to development of a pragmatic turbulent heat transfer model to be used in engineering applications. From these models, the AHFM as documented in Ref. [20] is considered one of the most promising. However, even this model still needs further development. Figure 2

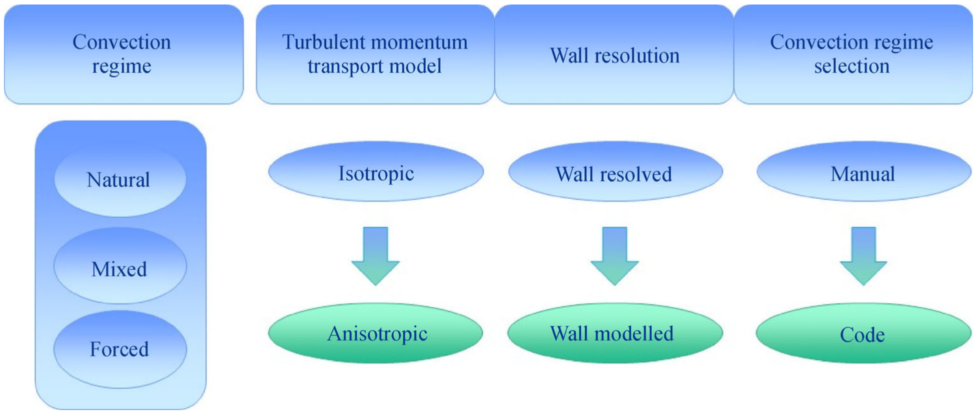

Fig. 2 Turbulent heat transport modeling.

summarizes the various modeling aspects that have to be considered in the development of a turbulent heat transport model. First of all, the model should be able to deal with all convection regimes. Secondly, since it is known that the flow in nuclear reactor fuel assemblies and pools is highly anisotropic, the turbulent momentum transport model should ideally be anisotropic. In the third place, the model should not only provide good results for computationally expensive wall resolved meshes, but also should provide reasonable results for wall modeled meshes which are used in most engineering simulations. Finally, the model should be able to select the suitable approach for the various convection regimes without interaction of the user, since, e.g., in large liquid metal pools, the various convection regimes may occur simultaneously. As mentioned before, even a promising model like the AHFM-NRG still needs further development since, although it can deal with all convection regimes and it has been implemented for an isotropic as well as an anisotropic momentum transport model (be it in different codes), it is now only available for wall resolved meshes and the user needs to specify the convection regime manually prior to the start of the simulation. The latter two challenges are the subject of the new European Horizon 2020 project PATRICIA which will run from 2020 to 2024.

Development of an advanced turbulent heat transfer model like the one discussed above, requires reference data to determine the model parameters and formulation. Apart from that, the reference data serve as the validation base for the model. Nowadays, a reference database is often a combination of experimental data and high-fidelity numerical data. Within Europe this database started to be build-up prior to and during the ASCHLIM (Assessment of Computational Fluid Dynamics Codes for Heavy Liquid Metals) project. After ASCHLIM, the database consistently expanded during consecutive projects. In fact, within the recent SESAME and MYRTE projects so many new data have been generated. However, it was decided not to expand the database further in the PATRICIA and PASCAL projects, but rather to focus on the exploitation of the existing database. A recent overview of the available database for liquid metal heat transfer CFD model development is provided in Ref. [19]. The largest part of the available data has been generated in the European context, however, the US colleagues and more recently the Chinese colleagues are also adding to this database.

2.2 Core thermal hydraulics

Core thermal hydraulics ranges from the analyses of a single sub-channel to the complete core and its bypasses including the inter-wrapper flow. An important part of this field is the thermal hydraulic assessment of a fuel assembly. On the status of fuel assembly thermal hydraulics for liquid metal cooled reactors, Ref. [21] and, specifically for wire wraps, Ref. [22] provide an elaborate overview on the ongoing efforts internationally. Figure 3 summarizes the various topics that need to be studied. The field of fuel assembly thermal hydraulics can be split into normal conditions and off-normal conditions.

Let's first consider normal conditions. In the past, most studies were restricted to fluid flow and heat transfer analyses of an assembly as it was envisaged on the drawing board. Indeed, this is the starting point of all analyses. A large number of experimental and numerical reference data have been created over time to allow development of engineering models to predict pressure losses and temperatures in the core, ranging from system thermal hydraulic codes to sub-channel codes, coarse-grid CFD, and finally well resolved CFD simulations. More recently, flow induced vibration in fuel assemblies is also being considered. On the other hand, reactor designers realize that the fuel assembly, as presented on the drawing board, differs from reality, since the fuel assembly will deform under the influence of operating conditions such as temperature, but even more so under the influence of irradiation. Fuel pins or assemblies may bow, swell, or shrink. In addition, the coolant chemistry might influence

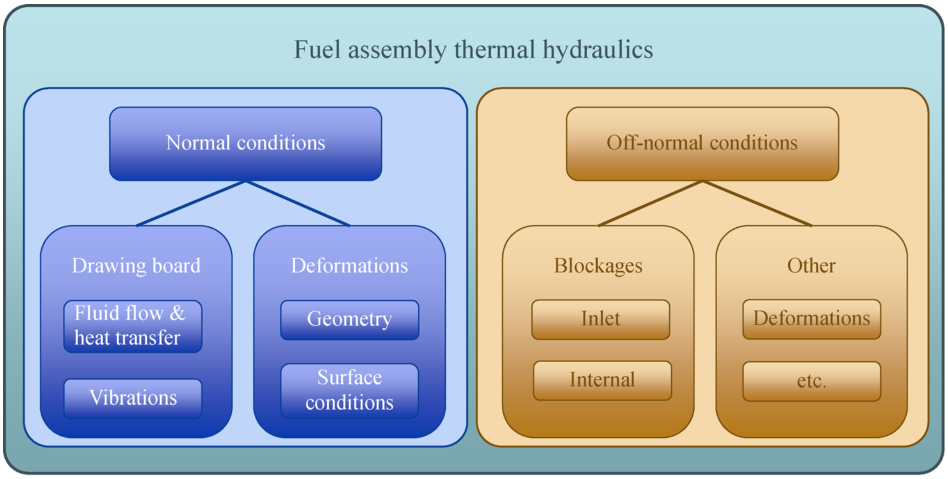

Fig. 3 Overview of fuel assembly thermal hydraulics.

the flow and heat transfer either by surface effects or by changes in the bulk properties of the coolant.

Let's now consider off-normal conditions. Although many challenges are in place, an important challenge for off-normal conditions is the fuel assembly behavior during a fuel assembly blockage. In fast reactors, the fuel assemblies are mostly closed (i.e., a housing or wrapper envelopes the fuel pins of one assembly and separates them from the next assembly). An important strategy in case of fuel blockage for such assemblies is to limit the effects to one assembly so that no propagation of the damage occurs. Blockages may occur at the fuel assembly inlet as well as internally. The size of the blockage may be such that it blocks the complete flow path or only a limited number of sub-channels. Different types of blockages need to be studied to learn about their potential consequences, to study ways of mitigating the associated issues, to learn about the possibilities of detecting the blockages, and to analyze possible ways of preventing such blockages.

Most liquid metal cooled reactors employ wires wrapped around the fuel pins as pin spacers. However, reactor designs employing grid spacers are also being developed. For both types of assemblies, a lot of work has been done in recent years. As mentioned before, Ref. [22] provides an overview of recent developments for wire-wrapped fuel assemblies. It provides a summary of experiments and numerical reference data and the comparison of engineering CFD models to these data in terms of accuracy. Apart from that, it shows that work is ongoing with respect to vibration assessment, both experimentally as well as computationally. To this respect, the experiments of Ref. [23] in a full 127-pin scale wire-wrapped fuel assembly in lead-bismuth have provided promising information to the designers that vibrations, if occurring, are not significant and are unlikely to cause vibration-induced fatigue damage. Next to the vibration frequencies obtained from the experiments, information on loads on the pins could be derived from fluid-structure interaction computations and provided to material specialists for the investigation of fretting. These computational attempts to simulate vibrations in a wire-wrapped assembly are expensive and complex, even though lots of progress has been booked in recent years, both in simulations in which the wires are taken into account implicitly, as well as in simulations in which the wires are explicitly modeled. The topic of deformations has received less attention, although Ref. [22] mentions some early works. These works deal with assessing the consequences of deformed fuel assemblies, whether it is the housing or the pins themselves. However, an attempt to predict deformations is also reported. The influence of surface conditions on the other hand, has hardly been studied. Geometrical deformations are to be studied in the new collaborative European PATRICIA project. In this project, it is envisaged to create new experimental data for predetermined deformations and use the measurements as input to designers and to modelers to validate their computational approaches. With respect to blockages, inlet blockages have mainly been studied numerically, but internal blockages have been studied numerically as well as experimentally. In experiments, conservatively, mostly solid blockages have been used so far. However, since simulations show large potential influence of leak flows, and since porous blockages are more likely to occur, solid blockages are probably too conservative, and a new experimental campaign is planned within the PATRICIA project in which porous blockages will be investigated.

With respect to grid-spaced liquid metal fuel assemblies, much less work has been done in recent years. The starting point is to analyze a fuel assembly from the drawing board, using a combination of experimental and numerical reference data. Such data can be exploited to validate more pragmatic numerical approaches. Besides, vibrations are being studied. First assessments have been done, but new experimental studies and numerical work are foreseen in the PASCAL project. Moreover, activities on deformed bundles will be initiated in the PASCAL project. For a grid spaced bundle, blockages are most likely to occur at spacer positions. Such blockages have been studied experimentally and numerically in the SESAME (Thermal-hydraulics Simulations and Experiments for the Safety Assessment of MEtal cooled reactors) project. Further studies will be required in the future to explain the discrepancies observed between both approaches.

As may be clear from the activities described above, for both grid-spaced as well as wire-wrapped liquid metal rod bundles, an indispensable step in numerical model development is to generate experimental reference data complemented by high fidelity CFD. Once these data are available, pragmatic engineering models can be validated and subsequently applied to a full-scale fuel assembly as demonstrated in Fig. 4. In first instance, the nuclear engineers rely on the models that have been validated based on assemblies as they are on the drawing board. However, additional validation is required with relevant reference data to fully trust the pragmatic engineering models for the various other configurations. The experiences of the simulations on solid blockages have shown clearly that one should not rely on the first validation program based on drawing-board configurations solely.

Apart from the cooling of and in a fuel assembly, the flow in the small spaces between two fuel assembly housings (or wrappers) is also being studied. This so-called inter-wrapper flow may significantly contribute to the cooling of fuel assemblies in certain reactor transients. Such work has been reported in Ref. [24]. The experiments performed within this campaign and the good comparison of the numerical approaches has led to the decision to prioritize other topics above the topic of inter-wrapper flow in the upcoming projects.

Together with the progress in the analysis of the cooling of a fuel assembly, the increased knowledge on inter-wrapper flow and the appropriate modeling of it culminated in a demonstration exercise on a fictive core design inspired by the FFTF core with subassemblies resembling the Superphenix design. For this analysis, a sub-channel code was used to model the behavior in the fuel assemblies, whereas a CFD code was used to model the behavior in the inter-wrapper flow.

2.3 Pool thermal hydraulics

The understanding of the thermal hydraulic phenomena occurring in the upper and lower plena of liquid metal cooled reactors is a critical issue in the design process. Knowledge of the convection patterns and thermal mixing and stratification in operational and accidental conditions is a necessity. These phenomena are highly three-dimensional and design-dependent. Experimental testing is, therefore, required. However, with the increase in computer power, 3D CFD simulation of these plena is nowadays feasible. In such CFD approaches, components which drive the flow in a liquid metal reactor pool, like the core, the heat exchangers, and the pumps, generally are not modeled explicitly in full detail but are represented by lower-order models such as porous bodies or momentum sources. These, of course, have to represent the component sufficiently accurately and need to be validated with experimental data. Once this has been achieved, the reactor designer can use CFD simulations to investigate various design options which would be costly to investigate in an experiment. Hence, experiments and simulations are increasingly being used in a complimentary way. Since liquid metals complicate accurate flow measurement, designers often start with water as simulant fluid for liquid metal. A large number of such experiments has been reported in Ref. [25]. However, to characterize the temperature field correctly, water is not very well suited because of its low conductivity. Complementary liquid metal experiments have to be conducted, providing design data to design engineers and validation data to the simulation engineers.

In the European collaborative projects, a range of experimental facilities has been used over the past couple of years to serve this purpose. The smallest facility is ironically called TALL-3D [26]. This facility includes a small three-dimensional pool embedded in a rather tall liquid metal loop system. In this small pool-type test

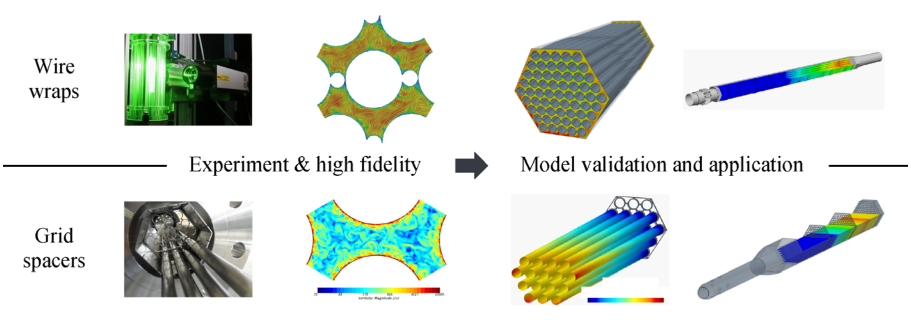

Fig. 4 From experimental and numerical reference data to pragmatic engineering model validation and application.

section, thermal stratification and mixing phenomena are studied. CFD validation and an elaborate sensitivity analysis are reported in Ref. [27]. This analysis shows that the boundary conditions, such as the liquid metal mass flow rate, inlet temperature, and heater power, followed by the turbulent Prandtl number and material properties (e.g., density and heat capacity of LBE) represent the major sources of modeling uncertainty. Large scale experiments have been performed in the Italian CIRCE (Circulation eutectic) facility using the integral circulation experiment (ICE) test section which has been instrumented such that relevant data for thermal stratification and flow patterns could be extracted [28]. CFD modeling is reported in Ref. [29] and shows that CFD allows to reproduce the general flow and temperature patterns in CIRCE in nominal and transient conditions with a reasonable accuracy. Prediction of the stratification in the pool was found to be sensitive to the modeling of the heat transfer from the core simulator and core outlet region to the large pool. From the small-scale TALL-3D and the large-scale CIRCE facilities, a next step has been taken with the prototypical European scaled pool experiment (E-SCAPE) in Belgium. E-SCAPE is a 1:6 scale representation of the MYRRHA reactor design [30] and, as MYRRHA, employs lead bismuth eutectic (LBE) as coolant. This facility, shown in Fig. 5, has been constructed with the objective of producing design information to the reactor designers, while at the same time, providing CFD-grade validation data for simulation engineers. The design of the facility has been supported largely by CFD simulations allowing proper positioning of instrumentation to characterize all necessary initial and boundary conditions, as well as sufficient flow parameters to make good comparisons. A first validation of CFD approaches is reported in Ref. [31], which shows that independent CFD teams, using different numerical tools and models, are able to predict temperature profiles that are in good agreement with the measurements in the LBE pool. On the contrary, pressure drops have been over-predicted by at least 15% and showed strong differences in this exercise. Subsequent investigations have revealed that mesh resolution and topology as well as proper momentum turbulence model are important factors, but especially a detailed and correct representation of the certain geometrical features and bypass flows [32]. Simplifications in the geometry which in first instance seemed justified (e.g., neglecting small gaps and rounded edges) have been found to be the cause of significant differences in the pressure drop. Further experiments (different transients focusing on natural circulation and the transition from forced to natural circulation) and validation of CFD models is foreseen using the CIRCE and E-SCAPE facilities in the upcoming PATRICIA and PASCAL collaborative European projects.

Taking profit from the experiences gained in modeling the experimental facilities described above (see Fig. 6), complete CFD models have been developed and are applied to the full-scale ALFRED design [33] and the MYRRHA [34] reactor design. These simulations aim at the assessment of the designs at a relatively early stage, when at the same time sufficient details are known. Full-scale three-dimensional analyses are supplementing the traditional system thermal hydraulic simulations which typically lack three-dimensional information. In addition, the full scale CFD simulations reduce the necessary experimental efforts in either water models or liquid metal models. In Ref. [35], it is explained that two full-scale CFD models have been constructed for the ALFRED

Fig. 5 Top view of the E-SCAPE experimental vessel.

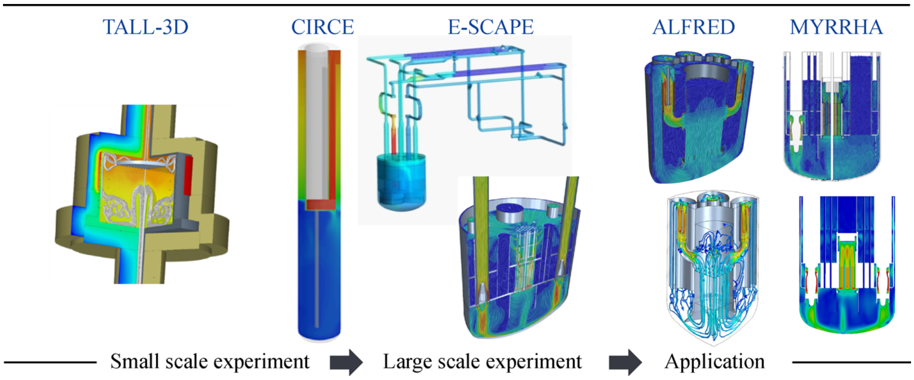

Fig. 6 Liquid metal pool thermal hydraulics CFD simulations from small scale to large scale experiments leading to full scale application.

design. These models have been used to investigate the effects of various design options without having to construct a physical model of the reactor. More advanced full-scale CFD models have constructed for the more mature design of the MYRRHA reactor. Using two different numerical codes and taking two different modeling approaches for the various components in the reactor, both modeling teams are able to reach an advanced CFD model. The model described in Ref. [36] has first been calibrated using the information on the nominal state of the reactor. After that, postulated transients like a protected loss-of-flow and a rupture of the diaphragm which separates the cold plenum from the hot plenum have been studied. The model described in Ref. [37] represents a later design version of the reactor and is obviously also calibrated using the nominal conditions, after which again a protected loss-of-flow transient is analyzed.

2.4 System thermal hydraulics

For the safety analysis of any type of reactor, numerical methods that describe the behavior of the complete reactor system are essential. Traditionally, system analysis is performed with so-called lumped-parameter or system codes. A non-exhaustive list of codes applied in Europe and the USA is provided in Appendix B. These typically make use of 0 or 1-dimensional control volumes in which the flow is solved, in conjunction with (empirical) correlations to take into account friction and heat transfer. They allow to analyze the complete reactor cooling system (primary, secondary and/or energy conversion loops) and can also take into account the behavior of the neutronics in the core. Obviously, validation of such codes is important. Mostly, the validation relies on a combination of separate effect tests and integral, design specific experiments and/or real reactor data from prototype, test or demonstration reactors [38]. Extensive databases exist for water cooled reactors. However, for the application of liquid metal cooled reactors, such data are scarce. Therefore, in recent years development of this validation base for liquid metal cooled reactors has been the focus in the European collaborative projects, profiting from liquid metal facilities like TALL in Sweden, CIRCE and NACIE in Italy, and E-SCAPE in Belgium, but also from real reactor data stemming from the PHENIX end-of-life tests. From the early assessments of the applicability of the traditional 0 or 1-dimensional numerical codes, it becomes clear that in large pools, which are common in liquid metal reactors, 3-dimensional effects are important. Therefore, an effort is being made in Europe and worldwide to expand the capabilities of the traditional codes toward inclusion of 3-dimensional effects, either directly in the system thermal hydraulic codes or through the development of a multi-scale coupled approach between a system thermal hydraulic code and a CFD code. Obviously, it is exhibited in Fig. 7 that such methods require, after a set of

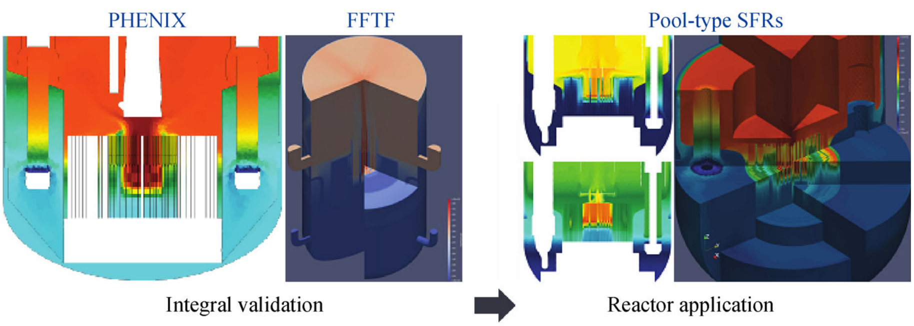

Fig. 7 Multi-scale simulation of liquid-metal reactor transients: integral validation on the PHENIX and FFTF reactors [39,40], application to new reactor designs [41].

verification tests and combined effect and large-scale tests, integral validation based on reactor scale data before they can be applied to new reactor designs.

When focusing on such multi-scale coupled approaches, one should realize that establishing a coupling between two independent numerical codes is far from straightforward. The developer has to make many choices in the development process which will influence the applicability and performance of the tool. In first instance, most developers focus on establishing a working coupling routine for the problem at hand. However, once such a coupling routine has been established, developers start to realize that the routine should be easily transferable to other applications. Therefore, the trend is now to develop generic multi-scale coupling tools that can deal ideally with different applications.

As mentioned before, concerning the coupling methodology, a lot of choices have to be made and many possible variants are being reported. The list below provides some examples, but is by no means exhaustive:

1) Determine qualitatively (via e.g., a phenomena identification and raking table) which local/3D phenomena may have a global impact, and thus justify a coupled approach.

2) From this analysis, partition the overall simulation into "domains" to be modeled at each scale.

3) Select the numerical codes best suited to model each of the domain, then decide whether the coupling routine should be integrated in one of the codes, or whether it should be stand-alone.

4) Select the coupling routine, e.g., making a choice between domain overlap and domain decomposition.

5) Implement data transfers between the codes at the "coupling boundaries" between the codes: these will usually correspond to the surfaces at the intersection of the calculation domains.

6) Develop a coupling algorithm in which these transfers will be used to ensure that, as time advances, the individual simulations performed at each scale are kept consistent with one another.

With respect to the decision on the coupling routine to be used, the following considerations can be made. On the one hand, a domain overlap approach is relatively easy to implement and robust, using an explicit numerical coupling scheme. On the other hand, this approach has a drawback related to mass conservation or pressure definition. A possible solution is to switch to a domain decomposition method. However, decomposition methods come with their own set of drawbacks: they are often less robust and thus require more complex implicit numerical coupling schemes. And even then, it may encounter stability issues. To overcome these, relaxation methods can be used, such as the Quasi-Newton method used by Ref. [42]. A decent and recent overview on existing approaches for system thermal hydraulic codes and multi-scale developments for liquid metal cooled reactors can be found in Ref. [11]. Further validation and development of promising multi-scale methods are foreseen in the PATRICIA and PASCAL collaborative projects.

2.5 Best practive guidelines (BPG), V&V and UQ

Design and safety analysis of liquid metal reactors relies on a combination of experiments and simulations. Since experiments are often hard, if not impossible to perform, the reliance on numerical simulations is large. An important aspect of every numerical simulation is the quality of the work performed. To guarantee the quality of the simulations, one should apply BPG and follow a V&V procedure along with UQ. Best practice guidelines basically consist of a set of advices or rules which the numerical engineer should apply or follow when performing a simulation. Such guidelines are widespread in the field of CFD, and where applicable, a CFD engineer is encouraged to apply them. However, the specific application of liquid metal thermal hydraulics asks for additional guidelines. One of the chapters in the recent textbook [11] on liquid metal thermal hydraulics, an important outcome of the SESAME project, specifically deals with BPG for liquid metal reactor applications. Besides, with respect to V&V and UQ, this textbook is a good reference. It clearly explains that verification provides evidence that the model is solved in a correct way, while validation provides evidence that the right model is solved. In addition, UQ covers a set of methods for the estimation of the accuracy of the model, taking into account the uncertainty in the model inputs and calibrated coefficients. In fact, this is not specific to liquid metal reactors. The explanation on V&V and UQ is generically applicable, though in Ref. [11] it is applied to liquid metal thermal hydraulic simulations. One of the main new developments needed, is the establishment of a practical V&V and UQ method for multi-scale coupled system simulations [43] like having been explained in the Section 2.4.

3 Summary and outlook

In this paper, the achievements and prospects with respect to liquid metal nuclear thermal hydraulics in the context of the developments at a European collaborative level have been summarized. European developments in the fields of heat transfer modeling, core, pool, and system thermal hydraulics have been presented. Even though significant progress has already been made in all these fields during the past 20 years, further developments have been identified. Part of these are the subject of two new European collaborative initiatives including thermal hydraulics topics, i.e., the PATRICIA and the PASCAL projects, which have started in the fall of 2020. The topics included in these two projects are: (further) development of pragmatic heat transport models for CFD; (further) analysis of the consequences of fuel assembly blockages; analysis of the effects of deformed fuel assemblies; extended validation of 3-dimensional pool thermal hydraulic CFD models; and (further) development and validation of multi-scale system thermal hydraulic methods.

In all cases, experiments and simulations are foreseen in parallel and supporting each other which is of utmost importance to realize the goals set out in the projects.

Appendix A

Table A1 introduces test facilities used in the European projects mentioned in this paper. For further information on these test facilities, the reader is recommended to consult the database from the IAEA [12].

Appendix B

Table B1 provides a non-exhaustive list of system thermal hydraulic codes used in Europe and the USA for analyses of liquid metal fast reactor systems. The reader is encouraged to visit the webpages of the various codes and consult descriptions in the open literature to obtain a better understanding of the applicability and features of the various codes. Short descriptions of ATHLET, CATHARE, RELAP5 Mod3.3, SAS4A/SASSYS, and SPECTRA can be found in Ref. [11].

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Table B1 Non-exhaustive list of system thermal hydraulic codes used in Europe and the USA for analyses of liquid metal fast reactor systems

System code	Developer(s)	Country	Na	PbBi	Pb
ATHLET	GRS	Germany	X	X	
ASTEC-Na	IRSN	France	X		
CATHARE	CEA/EDF/ Framatome/IRSN	France	X	X	X
DYN2B	Framatome/CEA	France	X		
RELAP5 Mod3.3/HLM	INL/UniPi	USA/Italy		X	X
RELAP5 3D	INL	USA	X	X	X
SAM	ANL	USA	X	X	X
SAS4A/ SASSYS	ANL	USA	X	X	X
SAS-SFR	CEA/JAEA /IRSN/KIT	n.a.	X		
SIM-SFR/LFR	KIT	Germany	X	X	X
SPECTRA	NRG	Netherlands	X	X	X
TRACE	US NRC	USA	X	X	X

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Table A1 List of test facilities used in the projects mentioned in this paper

Test facility	Country	Fluid	Heat transfer	Fuel assembly	Vibrations	Inter-wrapper flow	Pool	System
Low Prandtl gas loop	Belgium	HeXe	X					
DITEFA	Germany	GaInSn	X					
KASOLA	Germany	Na	X					
SEEDS	Netherlands	H_2O		X	X			
NACIE	Italy	PbBi		X				
KALLA	Germany	PbBi		X		X		
HELENA	Italy	Pb		X	X			
COMPLOT	Belgium	PbBi		X	X			X
TALL-3D	Sweden	PbBi	X				X	X
CIRCE	Italy	PbBi					X	X
E-SCAPE	Belgium	PbBi					X	X
FFTP	USA	Na						X
PHENIX	France	Na						X

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